

UNITED STATES ATOMIC ENERGY COMMISSION

REPORT ON THE
SL-1 INCIDENT, JANUARY 3, 1961

THE GENERAL MANAGER'S BOARD OF INVESTIGATION

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INFORMATION FOR PRESS AND RADIO
61-24 (SL-1)
Telephone JA 2-6640, Extension 217

FOR RELEASE
(In Newspapers dated Sunday,
June 11, 1961)

NOTICE TO EDITORS: Attached is the narrative section of the Board of Investigation's report presenting findings on the SL-1 (Stationary Low Power Reactor No. 1) accident which occurred last January 3 at the Atomic Energy Commission's National Reactor Testing Station near Idaho Falls.

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1. SUMMARY

A. Nature of Report

This report by the Board of Investigation is in response to the request of the General Manager of the Atomic Energy Commission to report on the SL-1 reactor incident. At the time of this writing (May 1961), there still remains substantial doubt concerning the initiating event causing the explosion within the reactor pressure vessel. The Board, therefore, feels constrained to restrict its observations concerning cause and responsibility to observable or demonstrable situations and events.

With this reservation, we present our findings at this time.

This report summarizes the current information before the Board pertaining to the circumstances surrounding the explosion on January 3, 1961, within the reactor vessel of the SL-1 (ALPR) reactor plant. Prior to the incident, there appear to have been a continuing deterioration of the burnable poison strips within the core and a worsening of the scram performance of the control rod system, neither of which circumstances necessarily was directly related to the incident. The evidence strongly indicates a nuclear incident of 50 megawatt-seconds, or more, which could credibly have been induced by rapid and extensive motion of the central control rod. There is no evidence to show that the actions of the operators on duty were in any way different than those prescribed and which had been carried out without incident many times before.

2. INTRODUCTION

A. Constitution of the Board

The General Manager, Mr. A. R. Luedecke, appointed a Board of Investigation on January 4, 1961, to investigate and report on the SL-1 reactor incident which occurred on January 3, 1961, at the National Reactor Testing Station (NRTS) in Idaho.[1/](#)

The Board first met during the evening of January 4, 1961, and has continued to perform its functions since that time. Its principal method of gathering information has been through the testimony of witnesses who appeared before the Board.[2/](#) The Idaho Operations Office, AEC, through its own staff, its Technical Advisory Committee, and its operating contractor, Combustion Engineering, Inc., has been the prime source of information and assistance to the Board.[3/](#)

The Board received additional technical advice and assistance from several observers who attended some of the sessions during which witnesses were interviewed.[4/](#)

B. The SL-1 Reactor

The reactor is a direct-cycle, boiling water reactor designed to operate at 3 Mwt gross capacity. The electric power and process heat were dumped to the atmosphere through load banks and heat exchangers, respectively. The reactor is fueled with enriched uranium plates clad in aluminum, moderated and cooled with light water in natural circulation.

The reactor vessel is 4.5 feet in diameter and 14.5 feet high.. It is surrounded by gravel on the sides and is supported on a concrete pad

resting on lava. The following equipment and components are located within the large silo-like structure: the reactor vessel, turbine-generator, heat exchanger and other water-handling components, air cooled condenser and fans and miscellaneous control equipment. The reactor control room is located in the adjacent support-facilities building. The reactor building was not designed as a leak-tight containment structure.[5/](#)

At 3 MWt power level, a saturated steam flow of 9000 pounds per hour was generated in the pressure vessel at 300 psig and 420 degrees F. About 85 percent of the steam was used to generate electricity. Fifteen percent of the steam by-passed the turbine into a heat exchanger, which simulated a space-heat load. The air-cooled condenser was used to reduce the requirement for water during plant operation.

A reference reactor core array of 40 fuel assemblies was designed. Channels were provided for a total of nine control rods: five 14-1/4 inch span cross rods and four T-shaped rods. In each rod, the cadmium absorbing section was 34 inches long, and with the rods positioned at indicated zero withdrawal, the cadmium overlapped the bottom and top of the active core by several inches. It was anticipated that the T-shaped rods would not be used in the reference 3 MWt core of 40 fuel assemblies, but that it might be desirable to use them in a full-size 59-assembly core. (The testimony indicates that the Argonne National Laboratory (ANL) was directed to develop a simple, small core and reactor system, but that to provide for flexibility

and possible increased performance demands, the extra fuel and control positions were included in ANL's design.) Originally, it had been intended to disperse a burnable poison, essentially in the form of boron, fully enriched in boron-10, in the fuel matrix. Because of developmental problems, not necessarily related to the boron in the fuel matrix, it was finally decided to expedite procurement of fuel assemblies by omission of boron from the fuel matrix. The neutron absorber was introduced in the form of thin, flat plates, welded to one or both side plates of the fuel assemblies, as had been done in the Borax III experiment. The full length burnable poison strips, fabricated of X-8001 aluminum and highly enriched boron, were positioned in the core so as not to be adjacent to control rod channels. Additional half-length strips were also attached to the bottom half of the opposite side plate of the 16 fuel assemblies in the center of the core.

3. ADMINISTRATION OF THE REACTOR PROJECT

A. General

The SL-1 reactor, originally designated the Argonne Low Power Reactor (ALPR), was designed as a prototype of a low-power, boiling-water reactor plant to be used in geographically remote locations. A request for such a plant to be built by the AEC was made by the Department of Defense in a letter dated September 27, 1955. The development and final design of the plant were assigned by the Division of Reactor Development, AEC, to the Argonne National Laboratory, to achieve an early operational version of this

type of plant.^{6/} Pioneer Service and Engineering Company was the architect-engineer and the construction was started by the Fegles Construction Company in July 1957. The design and proposed operation of the reactor ^{7/} ^{8/} were reviewed in February 1958 by the Hazards Evaluation Branch of the AEC's Division of Licensing and Regulation and also by the Advisory Committee on Reactor Safeguards. Approval was given by both of these groups for operation of the plant, as designed, at power levels up to 3 MWt. The AEC staff report stated "when higher power-level operation is contemplated, a report of additional hazards and consequences of operation at this new power level should be submitted together with a report of the operating experience at the 3 MW level."

B. Argonne National Laboratory

Argonne's role, under contract with the Division of Reactor Development, included the design, test and initial operation of the reactor plant. This work was carried out between 1955 and February 1959. Initial critical operation took place on August 11, 1958, and test operations culminated in a 500 hour run which terminated in December, 1958. Argonne's official role ended on February 5, 1959, when Combustion Engineering, Inc., assumed contractual responsibility for the plant. While Argonne has had no official responsibility since this time, its employees have, on several occasions, visited the reactor site to observe fuel inspection or have otherwise reviewed plant performance.

C. Combustion Engineering, Inc.

Combustion Engineering, Inc. (CEI) was not involved in the design, construction, or initial operation of the SL-1 reactor. CEI was involved with later operation of the reactor, in modifications to the reactor facility, and the continuation of training of military personnel. Military personnel have been on the site since 1958 for on-the-job training. Combustion Engineering personnel have been on the site since December, 1958. The contract between CEI and the AEC is for the term between December 14, 1958 and September 30, 1962.[9/](#) It is a cost-plus-a-fixed-fee contract for operation of the reactor and for the performance of research and development work at CEI's plant in Windsor, Connecticut. The contract contains a standard AEC clause concerning Safety, Health and Fire Protection.

This contract is administered by the Idaho Operations Office, AEC, with the day-to-day administration being carried out by the Military Reactors Division of that office.

CEI was responsible for the actual operation of the SL-1 reactor, for the routine training of military personnel and for developmental research programs.

The Contractor provided at the site a Project Manager, Operations Supervisor, a Test Supervisor and a technical staff of approximately six personnel. In recent months, the Project Manager spent approximately half time at the site and half time at the contractor's office in Connecticut. In

his absence, either the Operations Supervisor or the Test Supervisor was assigned as the Project Manager.[10/](#)

It was recognized that this situation was a temporary one, in that it was contemplated that a full-time, resident project manager would be assigned by CEI to the SL-1 plant. In discussion of the candidates for this position, and the necessary qualifications of a candidate, there was considered the existing arrangement whereby military personnel were not directly supervised (by personal, direct observation) during routine plant operation. Since early plans for operation of the SL-1 did not include any plans for any significant development work, the general plan for operation of the SL-1 was to utilize a military staff, comparable to that to be provided for a remote site, for the actual operation of the plant, with on-site supervision above the level of the plant superintendent, and general supervision assigned to the contractor. Because of the vacant position and because of the recent addition of some development work with the SL-1 plant (including the PL-1 condenser test, which required operation at higher power), the CEI "part-time project manager" wrote a letter to the AEC Contracting Officer's Representative, dated November 29, 1960, requesting written confirmation of the oral agreement that CEI shift supervisors were not required for routine supervision of plant operation during the night shifts. It was understood, as indicated by testimony before the Board, that CEI would provide supervision on any shifts when non-routine work was carried out. Further, the operating staff was

encouraged to - and frequently did - contact off-duty CEI supervisors if any unusual events or unforeseen circumstances arose when CEI supervision was not present. Testimony before the Board indicated that such an oral agreement did exist (although the letter had not been answered at the time of the incident) and that CEI did not believe there was any specific need for this supervision, from a safety standpoint, but that the broadened scope of the developmental program with the SL-1 plant suggested reconsideration of this working arrangement, including safety aspects. CEI did suggest that there was enough developmental work on site that CEI supervision might be regularly assigned. Agreement not to do this reflected an AEC decision not to push forward the developmental program with high priority. The testimonial record also indicates that the AEC's Idaho Office and the Army Reactors Office clearly believed that addition of night supervisors when only routine work was involved would defeat a part of the purpose of operating the reactor under the existing arrangement, i.e., to obtain plant operating experience with only military personnel.

A complete technical review of the reactor and its proposed operation was made in February 1959, when Combustion Engineering, Inc. became the contractor, by a Nuclear Safety Committee composed of personnel from the Connecticut offices of Combustion Engineering. It appears that no other such review or appraisal of the safety of reactor operation has been made since that time by the Combustion Engineering, Inc. Reactor operating

procedures, completely satisfactory to the AEC, have never been completed by Combustion Engineering, Inc., although they have been in the process of preparation and revision since mid-1959.

A reactor safety committee existed at the plant site. Its members included the CEI Operations Supervisor, the Test Supervisor, the Health Physicist and the Assistant Operations Supervisor. The Test Supervisor testified that the committee reviewed proposed test procedures and new operating procedures, but did not routinely review reactor operating experience or procedures unless specific problems were brought to it. They did not make any over-all comprehensive safety review of operations.

The proposed plans for operation of the SL-1, and the procedures for such operation, were subject to review and approval by the Director, Military Reactors Division, ID. The Contractor has routinely and consistently forwarded reports of reactor operations, including malfunction reports, to the Military Reactors Division. The Director of this Division, and more often the SL-1 Project Engineer on his staff, made frequent visits to the facility.

Regular written reports of reactor operations were forwarded to the Army Reactors Office, Division of Reactor Development, Hq. Periodic appraisals, through visits to the facility, of the safety of the SL-1 plant by members of the ID staff, did not include inspection of the nuclear safety of reactor operations. Trip reports by members of the Army Reactors Office,

Headquarters, especially during early operation of the plant, did include specific comments and recommendations concerning the operating procedures and a number of facility components at that time.^{11/} Quarterly review meetings, which dealt with reactor operational experience as well as programmatic plans, were attended by Army Reactors Office personnel as well as the ID personnel.

During a general Headquarters appraisal of ID contract administration, in 1959, assurances of ID reactor safety surveillance, including the SL-1 reactor, were obtained. Independent, validating review, by the Headquarters staff, of the ID reactor safety review system was not performed. There does not appear to have been a clearly defined requirement for this type of appraisal.

D. Department of Defense

Although the SL-1 reactor was a part of the program of the Army Reactors Branch, Division of Reactor Development, AEC, for the development of water reactors for military applications, the Department of Defense did not have the responsibility for this reactor, either under license or as a result of transfer of the reactor from the AEC according to the provisions of section 91b of the Atomic Energy Act. Military personnel at the site were either in training or a part of the cadre operating the reactor under the general supervision of Combustion Engineering, Inc. The plant superintendent, the chief operators (who also were shift supervisors), the

qualified operators and trainees were military personnel who operated the plant around the clock according to the procedures and policies provided by the contractor.

E. Atomic Energy Commission

Within the AEC the line of management responsibility for the SL-1 project is from the General Manager to the Assistant General Manager for Research and Industrial Development, to the Director, Division of Reactor Development, to the Manager, Idaho Operations Office (ID), to the Military Reactors Division, ID. Details concerning the definition and delegation of responsibility are given in Annex G.

At the Idaho Operations Office, the Director of the former Division of Military Reactors administered the CEI contract. A reactor engineer on his staff served as project officer for the SL-1 reactor.

Responsibility for safety of reactor operations was shared by each level of the line organization according to its function. Detailed delegation of this responsibility is not spelled out, although Manual Chapter 8401 does assign to the Operations Managers, and others, broad responsibility for assuring safety of reactor operations for those reactors under their contractual jurisdiction. (Evaluation of the hazards of specific reactor designs or operational programs by the staff of the Division of Licensing and Regulation (DLR) is not required, except as the Director of the Operating Division may specifically request. For new facilities, the Operating Division Director usually requests review by DLR before opera-

tion, although this is not required and there is no subsequent follow-up at the initiative of DLR. It is similarly not required that the Division Director get DLR review of later changes in a facility. This later review is often requested, but not as regularly as for new facilities. Later review was not requested for the SL-1. Inspection of reactor operations by the Division of Compliance is not required, but may be requested. Safety review and inspection by the AEC staff are required for all licensed reactors and for certain AEC-owned reactors.)

One area of apparent ambiguity concerning responsibility involved the Army Reactors Branch of the AEC. There was no functional statement (AEC Manual Chapter) for this organization, but a description of the duties of the Assistant Director for Army Reactors (approved by the General Manager on August 31, 1959), appearing on the organizational chart, states that the Assistant Director for Army Reactors "Plans and directs the joint AEC-DOD programs for the development of nuclear power systems to meet DOD requirements other than for naval vessel propulsion and for air and space vehicle applications," and that the Water Systems Project Branch "provides central management and technical supervision of the development, construction and operation of water systems reactors and plant prototypes. Provides direct supervision of work through Project Engineers, assigned individually by project, responsible for project management and continuous review and evaluation of contractor performance

and project progress. Prepares and maintains schedules, estimates, budgets, plans, correspondence, scope of work, and technical and operating data on all Branch projects. Assures the resolution of all technical problems that arise during the design, construction, testing, and operation of Branch Reactor projects."

The Army Reactors Branch also has a separate line responsibility under the Chief, Corps of Engineers, USA, for the Army reactor program, including, for example, the responsibility for the training program for military personnel and also the responsibility for the direction of a research and development program leading to the use of nuclear power plants at remote sites. Testimony from members of this office indicated understanding of the actual responsibility as follows: The Deputy Assistant Director for Army Reactors states "It is clearly understood. . .that we Of the Army Reactors were not authorized in our own name, as such, to direct changes to the contract or to direct operations, give direction to the Idaho Operations." The Assistant Director, in a prepared statement, states "As a staff member, I am charged with responsibility for planning, observing, advising appraising and recommending, but I have no direct authority over the operations of subordinate offices of the Division, nor can I give orders to officials in such subordinate offices".

Review by the Hazards Evaluation Branch of the Division of Licensing and Regulation was requested prior to operation, but not

subsequently. Review of the SL-1 project by the AEC's Advisory Committee on Reactor Safeguards was not required, but, on one occasion was requested by the Division of Licensing and Regulation prior to start-up of the reactor but not subsequently. Testimony indicated that Army Reactors personnel believed that requests for such reviews should be initiated by the field office. No requests for independent review were made after initial operation. (The testimony indicates that the loss of boron was well known within the Division of Reactor Development at AEC Headquarters, although it was not categorized as a serious condition in the reports transmitted to Headquarters. The difficulties with operation of the control rods appears not to have been known at Headquarters, and very little knowledge of the extent of the difficulty was known by the AEC staff at ID.)

4. OPERATING HISTORY OF THE REACTOR

A. General

The SL-1 achieved criticality on August 11, 1958, with ten fuel elements containing a total of 3.5 kg of U-235. There followed a series of critical experiments performed in the reactor, with and without poison strips, to determine the optimum fuel and poison loading to achieve the design objectives.[12/](#)

As a result of these critical experiments, a core was chosen with 40 fuel elements, forty full length and sixteen half length boron strips and five control rods. (Critical experiments were also performed on a full 59 element core that would have had higher power capability, but the design of such a core probably would have called for a different U-235 loading.) The differential and integral worth of the five control rods were obtained as a function of rod insertion into the core. Flux plots were made of the hot, zero power 40 and 59 element cores by use of irradiated gold and copper wires.

On October 24, 1958, the SL-1 achieved its full power rating of electricity and space heat. During October 29 - 30, 1958, a 40-hour xenon run was made. The SL-1 was then shut down and 8 hours later the reactor was brought to full power overriding peak xenon. There followed a 500 hour run at full power. The 500 hour run continued until December 11, 1958. The reactor was operated at a power level of 3 MW(th) up to November 1960. The plant remained shut down until March 6, 1959, for maintenance and inspection and for preparation of operating procedures and manuals. The Army Reactors Branch at this time stated that the procedures and manuals

turned over to CEI by ANL were not satisfactory for use by CEI. CEI was requested to prepare revised material. The material submitted by CEI was accepted as a basis for the start of reactor operations, but CEI was to further develop and modify the operating manuals and procedures after obtaining actual operating experience. Initial test operation by CEI, for the Windsor Nuclear Safety Committee, took place on March 6, and cold critical experiments began on March 30, 1959. The SL-1 was turned over to Combustion Engineering, Inc., for operation in February 1959.

A 1000 hour sustained power run was concluded in July 1959, and the plant then remained shut down for about a month for maintenance, modification and inspection.[13/](#)

Important shut-downs occurred in August, 1959, January, 1960, November, 1960, and December 23, 1960, to permit maintenance and inspection. Fuel elements were first removed from the core during September, 1959, and inspected by CEI and ANL personnel.

Subsequent inspections took place in October, 1959, August, 1960, and November, 1960. Initial discovery of the bowing of the boron strips, in the three inch sections between tack welds, was made in 1959. During the August, 1960, inspection it was observed that large amounts of the boron strips were missing from some fuel elements and the fuel elements in the center of the core were extremely difficult to remove, by hand. Removal caused plates to fall off and flaking of material. A considerable number of flakes were collected from the bottom of the vessel. As a result of these circumstances, it was felt that further removal of fuel elements might cause further loss of boron,

so that no further inspections were conducted. It was noted during the second periodic inspection in August, 1959, that the central fuel elements were difficult to remove.

B. Reactivity Changes

The design goal for the SL-1 reactor core was operation at design power level (allowing for normal outage) for a period of three years. The boron strips were incorporated in the core design to serve as a burnable poison, the depletion of which would compensate for the burning of fuel. Ideally, such an arrangement would lead to a constant reactivity value for the core (at operating conditions), which would be manifested by a nearly constant position of the banked control rods. The calculated reactivity behavior, in terms of banked rod position, vs. core exposure is given in Figure 1. Also plotted are the observed rod positions as a function of exposure. By 500 MWD, i.e., by May, 1960, it appeared that the core was gaining reactivity faster than predicted. In August, 1960, routine inspection of selected fuel elements revealed the extensive loss of boron. The large rate of gain of reactivity was ascribed to this boron loss.

Of greater safety significance (as opposed to interest in the core lifetime only), the greater rate of reactivity gain, and, in fact, the larger amount of reactivity gain, reduced the capability of the control rods to render the core subcritical (decreased the reactivity shut-down margin). Figure 2 indicates, as a function of core exposure, the banked rod position for different operating conditions. From these data, and from estimates of the worth of the control rods, estimates of the shut-down margin were made.

Because of the reduced shut-down margin, resulting from the boron loss, strips of cadmium were inserted in two of the T-rod control shrouds on November 11, 1960. The banked rod position, with the reactor cold, was determined at an exposure of 711 MWD, but not thereafter. The last part of the curve for the cold condition is an assumption of cold reactivity behavior, based on the observed behavior of the banked rods during equilibrium operation at 2.56 MWT. Thus, the effect of the cadmium at 2.56 MWT was observed to be approximately 1% in reactivity, and this was assumed to also be the case with the reactor cold.

CEI's estimate of the reactivity worth of the boron, at the beginning of core life, was 11%. A rough observation of a 2% gain in reactivity, over that predicted which was attributed to the loss of boron, led to the rough estimate that $2 \div 11 = 18\%$ of the boron originally present was missing from the core (this assumes uniform loss of boron from the core and certain other simplifying postulates concerning local reactivity effects).

Although numerical values for core reactivity, rod worth and shut-down margin are all subject to some uncertainty, in varying degree, depending on physical assumptions, the reactor condition, the calculational method or experimental technique, the available information indicates the following:

1. The initial shut-down margin for the cold reactor was probably somewhat less than intended - maybe approximately 3.5% δk actual margin versus an estimated 4-6% design margin. The actual margin was considered adequate.

2. The reactor could have been made critical by withdrawal of the central control rod only, anytime since startup or the reactor.
3. At the time of shut-down on December 23, 1959, the shut-down margin for the cold reactor was probably 2 to 3%, assuming rod worth was essentially unchanged from earlier measurements and calculations. With this assumption, and a similar one regarding rod #9 (the central control rod), criticality could be produced by withdrawal of this rod approximately 17 inches from the reference zero position.^{14/} Representative critical rod positions are given in Table 1 below.

Table 1
Representative Critical Rod Positions

Date	Core Exposure (MWD)	Conditions	Rods 1, 3, 5, 7 (Inches Withdrawn)	Rod 9
9/16/60	711	407 °F, zero power	14.2	14.4
9/16/60	711	2.5 MWt, no xenon	16.6	16.6
9/25/60	736	2.5 MWt, equil. Xenon	17.8	17.8
11/6/60	848	2.56 MWt, equil. xenon	17.6	17.6
11/15/60		Cadmium sheets inserted		
11/16/60	853	180 °F, zero power, no xenon	13.2	13.2
12/5/60	888	2.56 MWt, equil. Xenon	19.3	19.2
12/23/60	932	2.56 MWt, equil. Xenon	19.4	19.4

(In the initial critical experiments, with no boron present in the 4 x 4 array of fuel elements and with the side rods fully inserted, criticality was achieved with the central rod 14 to

14.5 inches withdrawn. The small size of this core would increase radial leakage, compared to that for a 40 element core, requiring greater withdrawal for criticality. The addition of cadmium and some boron in the actual core would increase absorption, requiring greater withdrawal for criticality, but would be at least partially offset by the presence of additional fuel. Eight additional bare elements, producing a 6 x 4 array, required insertion of the central rod from 14.5 inches withdrawn to 9.25 inches withdrawn to maintain criticality. These numbers serve to emphasize the uncertainty of the critical rod position in the absence of detailed knowledge of the composition of the core.)

C. Control Rod Drive Experience

From early operations onward, intermittent and increasing difficulty was encountered in the free movement of the control rods. At least over the first year of operations, and possibly in large measure thereafter, the difficulty arose from the abnormal performance of the seals through which the drive shafts penetrated the rack and pinion gear housings on top of the reactor. The rate of flow of seal water affected the performance of the rod drives, as did the presence of foreign matter. Increase filtration apparently reduced the problems associated with foreign matter. A study was in progress to seek an understanding of the variation of the scram performance of the rods, with seal water flow. This variation was not considered a serious problem, in that performance specifications were met, provided the seal water flow was at the design value. It was also stated that movements imposed in scram tests prior to reactor start-up and frequent

exercise of the rods seemed to improve rod performance, possibly by tending to clear out particles of dirt or rust in seals or bearings.

In more recent months, testimony before the Board and operating records indicate increased frequency of malfunctioning of the control rod drives. On the one hand it was postulated by several witnesses that the bowing of the boron strips attached to the fuel elements exerted sufficient lateral force to result in reduction of the clearance within the control-rod shrouds, restricting the free motion of the blades. On the other hand, several witnesses felt there was no evidence for such closing of the shrouds, but that there might be some accumulation of crud on the shroud and the blade surfaces; and that exercising the drives tended to prevent sticking of the rods in the shrouds. It was also indicated that the higher power operation, which took place only after November 1960, and the addition of the cadmium strips required further withdrawal of the control rods than had been previously required. Consequently, the drives were being used in a new region of the mechanical structure, where closer tolerances, or other differences, caused increased difficulties with rod motion.

The only known interferences within a shroud were:

1. A crimp or similar bend was observed in the top edge of the No. 1 shroud. A special stainless steel wedge-shaped tool was designed and used to straighten out this defect.

2. A dummy control blade, made of aluminum was fabricated for insertion and irradiation in the No. 4 shroud. On initial insertion, the blade could not be fully inserted. The wedge-shaped tool was used on this shroud also, but since it could not be inserted within the shroud, the actual remedy for insertion of the blade was to cut a portion off of the bottom of the blade.

After the incident a review was made of the Operating Logs from September 1, 1960, through December 23, 1960, by members of the Military Cadre. The data set forth in Annex J give all recorded examples of control rod performance.

According to testimony presented before the Board, all orders in the Night Order Book, for the instruction of reactor operating personnel, are given by either the Operations Supervisor, or the Plant Superintendent with the Supervisor's or Assistant Supervisor's concurrence, and the following orders reflect the efforts of the operations group to maintain the rods in an operable status by frequent exercise: 12/20/60, by the Plant Superintendent -

"Each shift will perform a complete rod travel exercise at approx. 4 hours after the start of shift. This rod exercising will be required of each shift until further notice."

12/21/60 by the Operation Supervisor

"Perform a complete rod travel exercise on the graveyard and subsequent shifts."

12/22/60, by the Plant Superintendent

"Do not perform control rod exercises during 2.56 MW power run." (Testimony indicates that a special power run to get equilibrium data was in progress at this time.)

A review of the Operating Log #13 reflects that the aforementioned orders were complied with by the operators. On December 23, 1960, when the reactor was secured, the Operating Log #13 includes, in part, the following:

"0825 Dropping rods to secure reactor

Rod drop times

#1 no drop

#3 dropped 1/2" and stuck

#5 clean drop in 0.82 sec.

#7 no drop

#9 clean drop in 0.81 sec.

"0827 Driving rods 1, 3, and 7 to zero

"0830 Controlling bypass steam flow to cool down to 2 °F/min.

"0835 Rod #3 dropped from 9" to 0.5 sec.

Rod #1 dropped from 16" to 9" in 1.3 sec."

Testimony indicates that this behavior was worse than usual, and that the Assistant Operations Superintendent remembered commenting that this was probably because of the preceding operation (with no rod exercising). The operating procedures

called for "scram-testing" the control rods before and during nuclear start-up of the reactor. Rods were dropped individually from a prescribed height before going critical, and also from another height after achieving operating temperature and pressure in the reactor vessel. Prescribed times for full insertion were given. If the prescribed times could not be met, reactor operation was not to proceed. Testimony indicates that if a rod did not meet the drop time criterion, the test was repeated.

Review of the experience with control rod performance indicates that this behavior was probably not as bad as had been experienced on some previous occasions, however. A complete record of performance, obtained from the operating logs, is attached as Annex J. The CEI Project Manager and the CEI Assistant Director of the Nuclear Division testified that they were not aware of my significant difficulty with the operation of the control rods and also were not aware of the entries in the log books over the past several months describing these difficulties.

5. SEQUENCE OR EVENTS SURROUNDING THE INCIDENT [15/](#)

After having been in operation for slightly more than two years, the SL-1 was shut down on December 23, 1960. It was planned that maintenance on certain components of the whole system would be performed during the succeeding twelve days and the reactor would again be brought to power on January 4, 1961. While maintenance work on several auxiliary systems of the plant was completed during this period, the only work planned for the reactor core was the insertion of 44 cobalt flux measuring assemblies into coolant channels between plates of the fuel elements throughout the core. Access to the core, to install these assemblies, through nozzles in the head of the reactor vessel required removal of the control-rod drive assemblies. This portion of the work was begun during the early morning hours of January 3, 1961. When the day crew (including personnel from the military and from Combustion Engineering) arrived at the SL-1 on January 3, disassembly had been completed. Installation of the flux measuring assemblies was accomplished during the day shift under the supervision of Combustion Engineering personnel.

The crew of the next shift (4:00 p.m. to midnight, January 3) consisted of three military personnel: the shift supervisor (a qualified chief operator), his operator-mechanic assistant (a qualified operator), and a trainee. This crew and the following one were assigned the task of reassembling the control rod drives

and preparing the reactor for startup.

First indication of trouble at the SL-1 reactor was an automatic alarm received at Atomic Energy Commission Fire Stations and Security Headquarters at 9:01 p.m. (MST) January 3, 1961. The alarm was immediately broadcast over all NRTS radio networks. At the same time, the personnel radiation monitor at the Gas Cooled Reactor Experiment gate house, about one mile distant, alarmed and remained erratic for several minutes.

Upon the receipt of the alarm, which could have resulted from excessive temperature, high radiation, by being struck by a missile, or a pressure surge in the region above the reactor floor, the Central Facilities AEC Fire Department at the NRTS and AEC Security Forces responded. A health physicist from the Materials Testing Reactor (operated for the AEC by the Phillips Petroleum Company) was called at this time.

Upon entering the SL-1 fenced area, the fire department personnel were unable to arouse the SL-1 crew. Access to the reactor support building was gained through use of the security patrolman's keys. The assistant fire department chief entered the reactor support building and immediately detected radiation levels up to 25 roentgens per hour (r/hr). He could observe none of the SL-1 crew in the reactor support building. The health physicist from the Materials Testing Reactor arrived and entered the reactor support building. He observed increasing radiation levels as he

proceeded toward the reactor building; he detected levels of 200 r/hr at the stairway to the reactor.

The decision to enter into the reactor building to attempt to locate the operating personnel was made after the arrival of the CEI plant health physicist. Entry by him, and others, located two of the crew on the floor near the reactor in a radiation field of approximately 1000 r/hr. One of the two crewmen was still living; the other, dead. Removal of the living man was accomplished by approximately 11:00 p.m. Shortly thereafter, he was pronounced dead by one of the AEC physicians who responded to the emergency call.

Subsequent entries were made over the next several days to remove the two remaining bodies and to recover certain equipment and records. Of over 100 people engaged in recovery operations during the first 24 hours after the incident and of the several hundred so engaged in the following week, 22 persons received radiation exposures in the range of three to 27 roentgens total body exposure. Precautionary medical check-ups did not disclose any clinical symptoms.

6. CONSEQUENCES OF THE INCIDENT

A. Injury to Personnel

The results of the post mortem examinations of the three deceased persons show that two of them died instantly as a direct or indirect result of blast damage and that the third man may have lived for about two hours after the incident. A fatal wound in the head of this third man precluded any possibility of survival. There was evidence of flash burns to limited areas of the bodies.

Personnel exposures, during the initial recovery operations are listed in the previous section. Since the removal of the third body, exposures to personnel engaged in the recovery operations have been limited to values less than normally allowed to radiation workers, i.e., less than 2.5 r per quarter.

B. Physical Damage

There appears to have been only minor physical damage to the reactor building. A buckling of reactor room ceiling directly above the reactor (the fan room floor) has been observed. Two of the shield plugs were driven upward out of the nozzles in the head of the reactor vessel and penetrated and stuck in the reactor room ceiling. One of these plugs was removed during subsequent operations. A peeling back of a portion of this ceiling indicates the possibility that some additional parts of the reactor system, for example, a shield plug, may have been projected into the fan room area.

Observations made with a pinhole camera for gamma rays indicated the presence of a high level gamma source in the fan room area (there is a possibility that what is being observed is gamma radiation emitted from the reactor but scattered from the structure above the reactor).

No conclusive evidence is yet available as to whether or not the reactor vessel itself has been damaged. Preliminary estimates have been made that the explosion may have caused an internal pressure as great as 500 psi, from observed damage above the reactor vessel and from calculations of energy needed to propel

certain components to observed locations. A portion of the sheet metal, covering some shield material on top of the reactor head, was bent upward, allowing dispersal of some of the gravel, steel punchings, and pelletized boron shielding material.

Photographs taken by movie and closed-circuit TV cameras have shown extensive damage to the core itself. The central control rod, No. 9, and a portion of its shroud appear to have been ejected completely from the core and are lodged below the central nozzle. Control rods Nos. 1, 3 and 7 appear to be within the core, though they may be displaced laterally and vertically to some extent. The shrouds of these control rods have been greatly distorted, and the top of the core is covered with debris from core components such as holddown plates and end boxes from individual fuel element assemblies. The core has been expanded, from internal pressure, to the point that it is in contact with the thermal shield near the wall of the reactor vessel at many points on its circumference removing the 6 to 9-inch clearance in the original core configuration. Two racks, those for Control rods Nos. 1 and 7, are protruding from their respective nozzles, though the threads on the ends of both appear to be damaged. The rack associated with No. 3 rod has been broken off near the upper surface of No. 3 nozzles.

The bell housing over control rod No. 5 rod extension had not been removed during the shutdown work and is still in place.

As a result of this, and because the position of rod No. 9 and its shroud have obscured vision, it is not possible to ascertain the position of this control rod.

The plate over the No. 8 nozzle through which the instrumentation leads from the core passed was blown from the nozzle, stripping the threads from each of the studs. The present location of the No. 8 plate is not known. Of the five shield plugs, only three have been observed, two in the ceiling of the reactor room (one of which was removed) and one lying on the top of the reactor head.

Thermocouple measurements and water-detecting probe measurements in the core have been made. Despite conflicting previous interpretations, it is now generally accepted that the level of the water in the reactor vessel, if indeed there is any water present, is at least 24 inches below the bottom of the active fuel. Since the first observations were made more than a month after the incident it is possible that what water was present just after the incident had evaporated before observation. Although there is no evidence to support it, and activity levels below the reactor vessel, would seem to indicate otherwise, it remains a possibility that the reactor vessel is cracked.

C. Nature of the Incident

In the absence of any direct evidence which would identify the initiating event, which resulted in the explosion within the SL-1 reactor vessel, the Board cannot state what actually

did initiate the incident. There appear to be several conceivable mechanisms, or sequences or events, that could have resulted in the observed effects. The relative credibility of these mechanisms is extremely difficult to establish without further information.

That an explosion took place is quite clear from the observed physical damage within and without the reactor vessel. Indications that a nuclear excursion took place were provided by the following: [16/](#)

1. Identification of the fission product yttrium-91 isotope, in a metallic sample shaken out of the clothes of one of the deceased.
2. Identification of activated copper (to Cu-64) in a cigarette lighter screw, belonging to one of the deceased.
3. Identification of activated copper in a watch band buckle, belonging to one of the deceased.
4. Identification of activated gold in a finger ring worn by one of the deceased.
5. Identification of activated Cobalt 58 in a gasket from the top of the reactor.
6. Identification of activated Chromium 51 in a gasket from the top of the reactor.
7. Identification of gross fission products in air samples taken one and two days after the incident.
8. Response of monitoring instruments at nearby sites to the passage of a radioactive cloud.
9. Observations of radioiodine contamination of sage brush.

Observed blast effects on equipment, components and personnel are not inconsistent with the conclusion that a nuclear excursion took place. That is, the energy release required to produce the pressures needed to cause the observed effects is comparable to that observed in the destructive BORAX experiment, on the one hand; and credible mechanisms and initial conditions can be postulated, on the other hand, that would lead to such an excursion.

The Board is aware of no chemical, metallurgical or physical analyses of any materials or components, the results of which would support the hypothesis of an initial chemical reaction which then induced a nuclear reaction by rearrangement of core components. In this regard, the Board has been advised that metallurgical examinations made after the incident probably would not establish conclusively whether a metal-water reaction initiated or remitted from a nuclear excursion.

D. Energy Release

One estimate of the energy release is based on the analysis of a metallic sample taken from the clothing of one of the deceased. This sample was analyzed for uranium isotopic composition, mass, and specific yttrium activity. This analysis, related (by assumption) to the total uranium present in the core, led to rough estimate of the total fissions during the excursion of 1.5×10^{18} equivalent to 50 megawatt seconds. It is believed that an energy release significantly less than this would not have produced the observed blast effects, and that an energy release greater by a factor of 3 or 4 would have produced much more drastic blast effects. Another estimate of the total energy release, based on analogy with SPERT experience, as well as observed atmospheric radioactivity, was a release as great as 500 megawatt seconds, indicating that there may have been more than one burst, or that there was additional lower power operation.

A number of estimates of integrated neutron flux have been made from the determinations of induced radioactivity in various samples (thermal neutron doses from 1×10^8 to 2×10^{10} nvt were

calculated). The extrapolation, to the number of fissions in a nuclear excursion is extremely uncertain; however, first, because the energy release (no. of fissions) is not large compared to the cumulative exposure of the core; and second, the unknown effects or shielding (water height, for example) and third, the unknown effect of delayed-neutron emitters released from the reactor vessel into the reactor room.

E. Activity Release

Aerial surveys conducted on several different occasions since the incident, at an altitude of 500 feet and above, have not indicated any activity levels (at the ground) greater than twice background levels. On the basis of meteorological information (inversion conditions, wind direction NNE at a velocity of 4 to 8 mph) and the observation of smoke plumes under similar conditions, together with air and ground samples, it appears that a narrow plume of gaseous fission products traveled SSW from the reactor building. Low-level off-site activity of sagebrush, due to iodine-131, was observed subsequent to the incident. Subsequent sampling in the immediate vicinity of the SL-1 facility indicated that low levels of gaseous iodine were released for a short period of time from the reactor or that iodine released at the time of the incident was undergoing translocation. As of April 7, 1961, measured I^{131} levels were essentially at background; close to the reactor building, soil samples did indicate a low contamination by strontium-90 for a period of time after the incident. Determinations of the strontium-90 content in five soil samples collected on

January 13, 1961 ranged from 1018 ± 18 d/m/20 gm near the Support Facility building, to 65 ± 8 d/m/20 gm approximately 20 feet east of the guard house along the perimeter fence. Intermittent radiation surveys in the vicinity of the SL-1 plant indicate that the gamma radiation has not decreased an appreciable amount. During the first week in February dose rates varied from the order of 10 r/hr, measured at the base of the reactor building, below the cargo door, to the order of 100 mr/hr, measured at a distance of approximately 300 feet from the reactor building

The implications of an SL-1 incident to the public in a populated area is discussed in a memorandum which is attached as Annex M.

7. POSSIBLE MECHANISMS FOR THE INCIDENT

From consideration of the factors which may have caused this accident, it is possible to conceive of several different items or combination of items which may have constituted the immediate initiating event. The accident could have occurred with no errors being committed on the part of the crew, though certain errors on the part of the operators also can be visualized as possible initiating events.

It is known that the tasks assigned to the operators (re-assembly of control rod drives) involved the lifting of the control blades. Testimony before the Board indicates that the Chief Operator and the Operator had performed this same task at least; four times before the occasion in question and that they had received specific training for this operation.

Their training procedure [17/](#) included the explicit instruction that during disassembly, the central rod was not to be raised more than four inches. The reason for this limit was not given in the procedure. From the position of the men after the incident and the injuries they suffered, we are unable to rule out the possibility that one, or possibly two, of them were engaged in lifting the central rod at the time of the explosion. At present, however, there is no direct evidence on this point.

In the light of measurements made prior to the reactor shutdown on December 23, 1960, it would have been necessary to raise the central control rod a minimum of 16 inches at that time to produce criticality. On the basis of existing Information on the reactivity worth of the central control rod (prior to shutdown) and the results of BORAX and SPERT experiments, [18/](#) it is estimated that this rod would need to be withdrawn another 6 to 8 inches at a rate of approximately 24 inches per second in order to produce a nuclear excursion of the magnitude estimated to have occurred. While these actions and conditions appear credible, they do not appear probable in the light of the evidence thus far available.

Additional factors can be considered at this time, which involve the possibility that some changes occurred in the properties of the reactor between December 23, 1960 and January 3, 1961 - changes which would minimize the capability of the central rod system to maintain the reactor shutdown. There is no direct evidence at present that any such changes

took place. If loss of cadmium or loss of boron did occur during the shutdown period in question, the shutdown margin of reactivity would have been reduced. With a reduced shutdown margin of reactivity, substantially less withdrawal of the central control rod would have produced criticality.[19/](#)

Other conceivable initiating events, though at the present their likelihood appears to be low, include: [20/](#)

- (a) A water-metal, hydrogen explosion, or other chemical reaction, below the reactor core, which would drive the central rod or several of the rods up out of the core, or that would lift the seal plugs and therefore the attached rods by a general pressure increase.
- (b) Addition of water to a core which had become dry and otherwise changed.

It should be emphasized that the foregoing discussion is limited to possibilities and is not intended to imply any degree of probability. It appears now that the most likely immediate cause involved some unusually large and rapid movement of the central control rod.

8. CONCLUSIONS

In the absence of additional information concerning the initiating event for the incident, the Board is unable at this time to be more specific about the nature, cause and extent of the incident.

- A. An explosion occurred in the SL-1 reactor at approximately 9:00 P.M., on January 3, 1961, resulting in the death of three persons, in damage to the reactor and to the

reactor room, and in high radiation levels (approximately 500-1000 r/hr) within the reactor room. On April 1, the levels had decreased to the order of 100-200 r/hr and were decaying with a half-life of approximately 40 days.

- B. Two members of the crew were killed instantly by the explosion. The third died within about two hours as a result of an injury to the head.
- C. The explosion involved a nuclear reaction. The thermal nvt above the reactor was estimated to have been approximately 10^{10} n/cm², and may have resulted from more than a single burst of radiation.
- D. Chemical and radioactivity measurements on a single fragment of reactor fuel ejected by the explosion, if representative of the total fuel, suggest that the reaction may have resulted in 1.5×10^{18} fissions. This would have produced 50 megawatt-seconds of energy. Other estimates, based on decay of gaseous activity and on analogy with SPERT and BORAX experimental results, give a range from 100 MW-seconds to 500 MW-seconds, for the total energy release.
- E. At the time of the explosion, the reactor crew appears to have been engaged in the reassembly of control rod mechanisms and housings on top of the reactor. The pressure generated within the reactor, which probably reached several hundred pounds per square inch, was

vented through a number of partially closed nozzles in the head of the reactor, blowing out shield plugs, portions of control rods, and some fuel.

- F. The explosive blast was generally upward from the ports in the top of the reactor. Structural damage to the building, principally due to objects projected from the nozzles, was slight. Damage to the reactor core is extensive, although there does not appear to have been gross melting of the uranium core.
- G. Some gaseous fission products, including radioactive iodine, escaped to the atmosphere outside the building and were carried downwind in a narrow plume. Particulate fission material was largely confined to the reactor building, with slight radioactivity in the immediate vicinity of the building.
- H. At this time it is not possible to identify completely or with certainty the causes of the incident. The most likely immediate cause of the explosion appears to have been a nuclear excursion resulting from unusually rapid and extensive motion of the central control rod. As yet there is no evidence to support any of several other conceivable initiating mechanisms.
- I. It is known that a variety of conditions had developed in the reactor, some having their origin in the design of the reactor and others in the cumulative effects of reactor operation, which may have contributed to the

cause and extent of the incident. Among these conditions were the loss from the core of the burnable boron and the condition of the control rods that caused sticking.

FOOTNOTES

- 1/ Copies of the teletypes concerning formation of the Board of Investigation area attached as Annex A.
- 2/ A list of witnesses, who appeared before the Board, is attached as Annex B.
- 3/ The membership of the Technical Advisory Committee is given in Annex C.
- 4/ A list of observers is given in Annex D.
- 5/ A series of photographs and drawings are attached as Figures 4 through 8.
- 6/ AEC Staff Paper AEC 420/27 Argonne Low Power Reactor Project, October 31, 1955.
- 7/ ALPR Preliminary Design Study, ANL-5566, April 1956.
- 8/ Hazard Summary Report on the ALPR, ANK-5744, completed October 1957, published November 1958.
- 9/ Pertinent contractual arrangements and agreements are given in Annex F.
- 10/ An organization chart for the CEI administration of the SL-1 plant is attached as Annex I.
- 11/ A summary of inspections and visits is attached as Annex E.
- 12/ Detailed test results are given in a report of a talk by D. H. Shaftman, on "Pre-Power, Zero-Power Reactor Physics-Experiments in the ALPR, Presented at ANPP Reactor Analysis Seminar, October 11, 1960" and "Initial Testing and Operation of the Argonne Low Power Reactor (ALPR)", ANL-6084, December 1959.

- 13/ A detailed chronology of reactor operation is attached as Annex H. A summary of equipment malfunctions is attached as Annex K.
- 14/ A representation of control rod worth is attached as Figure 3.
- 15/ A detailed chronology of events before and after the incident was contained in the AEC press release of January 12, 1961.
- 16/ Detailed results of activation data are attached as Annex L.
- 17/ A copy of the procedure is attached as Annex N.
- 18/ A discussion by a Board Consultant or considerations of rate of change of reactivity and total change of reactivity related to energy release is attached as Annex O.
- 19/ A discussion by a Board Consultant of possible reactivity additions, since construction, of the SL-1, is attached as Annex P.
- 20/ A discussion by a Board Consultant of the significance of chemical reactions in the SL-1 incident is attached as Annex Q, and a metallurgical evaluation of the SL-1 core components is attached as Annex R.